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Tom Tankersley
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Waterford 3

10CFR50.73(a)(2)(iv)(A)

W3F1-2005-0086

January 11, 2006

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: Licensee Event Report 05-005-00
Waterford Steam Electric Station, Unit 3 (Waterford 3)
Docket No. 50-382
License No. NPF-38

Dear Sir or Madam:

Attached is Licensee Event Report (LER) 05-005-00 for Waterford Steam Electric Station Unit 3. This report provides details of a condition involving a manual Reactor Trip due to lowering Main Condenser vacuum upon the loss of all running Circulating Water Pumps. This condition is being reported pursuant to 10CFR50.73(a)(2)(iv)(A) due to manual actuation of the Reactor Protection System and automatic actuation of the Emergency Feedwater System.

This report contains no new commitments. If you have any questions, please contact Oscar Pipkins at (504) 739-6707.

Sincerely,

Tommy E. Tankersley
Licensing Manager

TET/OPP/cbh

Attachment(s)

IE22

cc: Mr. Bruce S. Mallett
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U. S. Nuclear Regulatory Commission
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NRC Senior Resident Inspector
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NRC FORM 366 (6-2004)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104	EXPIRES: 06/30/2007

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE
Manual Reactor Trip Upon Loss of All Circulating Water Pumps and Lowering Condenser Vacuum

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME: NA	DOCKET NUMBER: 05000
11	11	2005	2005	- 5 -	00	01	11	2006	FACILITY NAME: NA	DOCKET NUMBER: 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: *(Check all that apply)*			
1	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
100	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Waterford 3 SES Oscar P. Pipkins	504-739-6707

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
X	KE	2	Eagle Signal Div	Y					

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On November 11, 2005, at approximately 20:34, with the plant operating at 100% power (Mode 1), Waterford 3 manually tripped the reactor due to lowering Main Condenser vacuum, caused by a loss of all Circulating Water (CW) Pumps. Subsequently Emergency Feedwater Actuation Signals (EFAS-1 and EFAS-2) were received due to low Steam Generator Levels. The plant was then maintained in Mode 3 with both Steam Generators being fed from the Auxiliary Feedwater System with Steam Generator levels in the normal operational band for Mode 3. Failure Mode Analysis identified a degraded timer relay in the CW Pump discharge valve control circuit as the most likely cause. The relay was replaced prior to plant restart. Emergency Feedwater actuated automatically. Adequate water level was maintained in the Steam Generator to ensure decay heat removal from the RCS. The event is not considered a Safety System Functional Failure. The event did not compromise the health and safety of the general public.

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REPORTABLE OCCURRENCE

On November 11, 2005 at 20:34, Waterford 3 manually tripped the reactor due to lowering main condenser vacuum caused by a loss of all Circulating Water Pumps [KE]. Subsequently Emergency Feedwater Actuation Signals EFAS-1 and EFAS-2 were received due to low Steam Generator levels actuating the Emergency Feedwater System [BA]. The condition was reported to the NRC Operations Center within eight hours. This notification was followed up with a corrected notification on November 14, 2005 after it was subsequently confirmed that the condition was reportable under both four and eight hour reporting criteria. The event was reportable within four hours under criteria 10CFR50.72(b)(2)(iv)(B) for a manual trip of the plant in anticipation of receiving a Reactor Protection System (RPS) [JC] trip with the Reactor critical. The condition was also reportable within eight hours in accordance with 10CFR 50.72(b)(2)(iv)(A) for the automatic actuation of EFAS upon low Steam Generator levels. The failure to report within four hours was entered in Waterford 3's Corrective Action Program (CR-WF3-2005-04599). The reactor manual trip is reportable in writing (Licensee Event Report) within 60 days in accordance with 10CFR50.73(a)(2)(iv)(A) due to the manual actuation of the RPS and due to the automatic actuation of the Emergency Feedwater system.

INITIAL CONDITIONS

Just prior to the initiating events, the plant was operating in Mode 1 at 100% power. There were no procedures being implemented specific to this condition. There were no Technical Specification Limiting Conditions of Operation specific to this condition in effect. Three of the four Circulating Water Pumps were running.

EVENT DESCRIPTION

The CW System provides cooling water to the Main Condenser to condense steam exhausted from the main turbine, the feedwater turbines [SJ], and condensate drain sources. The CW System also provides cooling to the Reactor Coolant System [AB] by cooling steam dumped to the Condenser through the bypass valves during plant startup and shutdown. The CW System contains four large pumps. Plant design allows for a minimum of 3 pumps operating during winter and four pumps operating during summer to obtain 100% power based on river temperature.

On November 11, 2005 at 20:34, while operating at 100% power with three of the four Circulating Water (CW) Pumps running, the "A" CW Pump discharge valve began to close and at 90% closure the pump tripped per design. Within 45 seconds, the "B" CW Pump tripped due to automatic discharge valve closure. At this time the Control Room Supervisor (CRS) directed removing 200 MWs of load from the Main Turbine [TA] at 20 MW per minute. While the Main Turbine was being adjusted for load removal, CW Pump "D" also tripped due to automatic discharge valve closure. The CRS directed a manual reactor trip due to no CW Pumps running and rapidly lowering Condenser vacuum. The reactor was manually tripped at 20:34 and Operations procedure OP-902-000, "Standard Post Trip Actions", was entered. Emergency Feedwater actuation signals EFAS-1 and EFAS-2 initiated due to low Steam Generator levels. The Operations crew entered Operations

Procedure OP-902-006, "Loss of Main Feedwater Recovery Procedure" at 20:45 because both Feed Water (FW) Pumps had tripped on lowering condenser vacuum. Subsequently, the Steam Bypass Control System [JI] was unavailable, by design, due to a condenser interlock. As a result of the full load rejection, steam generator pressure rose to the Atmospheric Dump Valve setpoint. During the initial pressure rise, the steam generator wide range and narrow range level indications deviated (believed to be due to velocity effects in the downcomer region) resulting in the Emergency Feedwater backup flow control valves opening prior to the Emergency Feedwater Actuation Signal. Normally, it is expected that the Emergency Feedwater Actuation Signal occurs first and then the flow control valves open. Limited, initial surveillance testing and engineering analyses confirmed that no malfunction occurred and the system responded properly for the sequence of inputs to which it was subjected. Steam Generator pressure was controlled by the Atmospheric Dump Valves, which were both set between 990 – 992 psig. The plant was maintained in Mode 3 with both Steam Generators being fed from the non-safety Auxiliary Feedwater [BA] System with Steam Generator levels in the normal operational band for Mode 3. The EFAS was reset. The plant was maintained in Mode 3 while a Post Trip Review was performed.

CAUSAL FACTORS

Results of condition investigations and failure modes analysis conclude that the most likely cause of the condition was a synchronous motor driven reset timer (CW-SDT) malfunction, causing the relay to drop out when not energized, which sequentially closed the CW Pump Discharge Valves at about one minute intervals and secured all running CW pumps. The timer relay was sent off-site (Southwest Research Institute) for evaluation, where it was determined to have extensive thermal degradation. Additional troubleshooting is being evaluated under the Corrective Action Program.

CORRECTIVE ACTIONS

Corrective actions taken include:

- CW synchronous motor driven reset timer was replaced prior to plant startup, and
- the reset timer was sent to Southwest Research Institute for failure analysis.

Additional appropriate corrective actions are being evaluated and administered under the Corrective Action Program (CR-WF3-2005-4593).

SAFETY SIGNIFICANCE

The loss of all circulating water pumps resulted in a loss of condenser vacuum. Operators took action to initiate a manual reactor trip prior to condenser vacuum dropping to the point at which an automatic turbine trip would occur. The operator actions mitigated the consequences of the loss of condenser vacuum.

FSAR Chapter 15.2.1.3 analyzes a loss of condenser vacuum. The FSAR analysis is conservative, intended to bound actual plant operation and assumes no Operator action for 30 minutes into the event. As a result, reactor trip occurs on high pressurizer pressure and the RCS safety relief valves open. The peak RCS pressure for the FSAR loss of condenser vacuum analysis is 2732 psia, which is below the acceptance criteria of 2750 psia.

The event on November 11, 2005 was significantly less severe than the FSAR loss of condenser vacuum analysis. Prompt Operator action to initiate a manual reactor trip prior to an automatic turbine trip on loss of condenser vacuum prevented any significant RCS pressure rise and the RCS safety valves from opening. Safety equipment responded to the reactor trip. Emergency Feedwater response was appropriate to ensure decay heat removal from the RCS. At all times during the transient, adequate water level was maintained in the steam generator to ensure decay heat removal from the RCS.

The impact on core damage frequency is negligible since the RCS safety valves did not open (there was no loss of RCS inventory) and multiple additional failures would be necessary for core damage to occur.

In conclusion, this event is bounded by the loss of condenser vacuum analysis in FSAR Section 15.2.1.3. Prompt Operator action to initiate a manual reactor trip mitigated the consequences of the transient to be significantly less severe than that presented in the FSAR. Therefore, the safety significance of this event is small.

This condition is not a Safety System Functional Failure. Per NUREG-1022, Event Reporting Guidelines, 10 CFR 50.72 and 50.73, there are four safety system functions: ability to shut down the reactor and maintain it in a safe shutdown condition, ability to remove residual heat, ability to control the release of radioactive material, and ability to mitigate the consequences of an accident. The subject condition involved a manual reactor trip and plant shutdown that did not involve a failure of the plant's ability to achieve the four safety system functions.

SIMILAR EVENTS

A search was performed for other similar reported events at Waterford 3. No similar events were identified.

Condition Reports for the Entergy Nuclear South plants dating from 1993 were also searched. No similar events were identified.

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ADDITIONAL INFORMATION

Energy Industry Identification System (EIIS) codes are identified in the text within brackets [ ].